November 26, 2002

Mr. A. Christopher Bakken III, Senior Vice President
and Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS (TAC NOS. MB3955 AND MB3956)

Dear Mr. Bakken:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 272 to Facility Operating License No. DPR-58 and Amendment No. 253 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated January 14, 2002.

The amendments would revise Unit 2, TS 3.4.2, “Safety Valves – Shutdown,” and TS 3.4.3, “Safety Valves – Operating,” to increase the allowable as-found setpoint tolerance for the Unit 2 pressurizer code safety valves from plus or minus ($\pm$) 1 percent (%) to $\pm$ 3%. In addition, the amendment would add an allowable $\pm$ 1% as-left setpoint tolerance for the pressurizer code safety valves to Unit 1 and Unit 2 TS 3.4.2 and TS 3.4.3.

A copy of our related safety evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 272 to DPR-58
2. Amendment No. 253 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

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**No legal objection, See editorial comments

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*Provided SE input by memo

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DATE	11/16/02	11/12/02	10/15/02	11/25/02	11/26/02

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INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 272
License No. DPR-58

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated January 14, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 272, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 26, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 272

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 4-4	3/4 4-4
3/4 4-5	3/4 4-5

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 253
License No. DPR-74

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated January 14, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 253, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 26, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 253

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 4-4	3/4 4-4
3/4 4-5	3/4 4-5

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 272 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 253 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated January 14, 2002, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant, Units 1 and 2. The proposed amendments would revise Unit 2 TS 3.4.2, "Safety Valves – Shutdown," and TS 3.4.3, "Safety Valves – Operating," to increase the allowable as-found setpoint tolerance for the Unit 2 pressurizer code safety valves from plus or minus 1 percent to plus or minus 3 percent. In addition, the amendment would add an allowable ± 1% as-left setpoint tolerance for the pressurizer code safety valves to Unit 1 and Unit 2 TS 3.4.2 and TS 3.4.3.

2.0 REGULATORY EVALUATION

The Nuclear Regulatory Commission (NRC) regulatory requirements related to the content of TSs are set forth in 10 CFR 50.36. Specifically, 10 CFR 50.36(c)(2)(ii) sets forth four criteria to be used in determining whether a limiting condition for operation (LCO) is required to be included in TSs. These criteria are: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant system (RCS) pressure boundary; (2) initial plant conditions that are assumed in a design-basis transient and accident analysis; (3) components or systems that are used for mitigating consequences of the design-basis transient and accident; and (4) components or systems which probabilistic risk assessment has shown to be significant to public health and safety.

The Standard Technical Specifications (STS) were developed based on the criteria in 10 CFR 50.36(c)(2)(ii). Existing LCOs and related surveillance requirements (SRs) included as TS requirements which satisfy any of the criteria specified in 10 CFR 50.36(c)(2)(ii) must be retained in the TSs. The Nuclear Regulatory Commission (NRC) encourages the licensee to upgrade their TSs consistent with those criteria and conforming, to the extent practical and consistent with the licensing basis for the plant, to the current STS.

D. C. Cook, Units 1 and 2 use pressurized water type nuclear steam supply systems furnished by Westinghouse Electric Corporation. As noted above, since the STS were developed based on the criteria in 10 CFR 50.36(c)(2)(ii), the staff reviewed the proposed TS changes in accordance with NUREG-1431, “Standard Technical Specifications - Westinghouse Plants.” Pressurized safety valves (PSVs) are part of the primary success path and are credited in the updated final safety analysis report (UFSAR) for the accident and safety analyses to mitigate the effects of the design-basis events. In accordance with the Criterion 3 of 10 CFR 50.36(c)(2)(ii) discussed in Section 2 above, a TS LCO is required for the PSV setpoints. Since the licensee proposed TS changes to increase the PSV setpoint tolerances, it provided a transient analysis to support the TS changes for the staff to review and approve. The staff reviews the transient analysis in accordance with Chapter 15 of the NRC’s Standard Review Plan (SRP), NUREG-0800.

3.0 TECHNICAL EVALUATION

The staff has reviewed the proposed TS changes related to the setpoint tolerances for PSVs and the associated supporting transient analysis for D. C. Cook Units 1 and 2, and prepared the following evaluation. The purpose of the staff review is to confirm that the licensee’s design-basis transient analysis is based on acceptable methods; the analytical results meet the required acceptable criteria; and the proposed TSs appropriately reflect the result of the acceptable analysis.

3.1 Setpoint Tolerance -- D. C. Cook, Unit 2

The main design purpose of the PSVs is to provide overpressure protection for the reactor coolant primary system. In assessing the effects of the TS changes on the design-basis transient analysis, the licensee evaluated the existing transient and accident analysis and identified that the events that resulted in the greatest increase in pressurizer pressure were: (1) reactor coolant pump locked rotor, (2) loss of external load or turbine trip, (3) loss of normal feedwater flow (LONF) and loss of all alternating current (AC) power to the plant auxiliaries (LOAC), and (4) feedwater line break. The consequences of these events were most sensitive to the proposed changes in valve setpoint tolerances. The licensee reanalyzed these events to assess the effects of an increase of ±3 percent in the as-found PSV setpoint tolerance on the results of the transient analysis of record.

3.1.1 Reactor Coolant Pump Locked Rotor

For the single reactor coolant pump locked rotor event, the reactor will be tripped on the low reactor coolant flow. The results of the departure from nucleate boiling ratio (DNBR) calculations are dominated by the calculated hot channel heat flux and the RCS flow during the transient. The changes in the as-found setpoint tolerance for PSVs affect the heat flux and the RCS flow negligibly, and thus, affect the DNBR analysis for the event insignificantly.

In addition, the licensee stated that the analysis of record assumed a lower initial pressure (2100 psig) for the DNBR calculation during a locked rotor event. Holding the pressure constant at the lower initial value results in a lower DNBR value and is a conservative assumption. Therefore, the staff concludes that the analysis of record for the DNBR calculation remains bounding and is acceptable for the locked rotor event to support the TS changes related to the setpoint tolerance for the PSVs.

The locked rotor event was reanalyzed for overpressure considerations. However, the case with a negative tolerance (-3 percent) for the PSV setpoint had no inadvertent effect on the transient since the opening of PSVs would provide heat sinks for removal of energy from the RCS. The use of PSVs with inclusion of a negative tolerance would result in an earlier opening of the valves and decreased the peak pressurizer pressure during the transient and was therefore, not reanalyzed. For the overpressure reanalysis, the licensee assumed that the PSVs would open only when the calculated pressures reached values corresponding to the specific valve setpoints with the associated tolerance of +3 percent. The licensee’s reanalysis was performed with the NRC-approved LOFTRAN code. The initial conditions were assumed to be the same as those currently assumed in the licensing-basis analysis. The results of the reanalysis showed that the peak pressure is below 110 percent of the design pressure.

Since the licensee used current licensing-basis computer code to analyze this event and showed that the peak RCS pressure would remain within 110 percent of the design pressure, the staff concludes that the reanalysis of the reactor coolant pump locked rotor event meets the acceptance criteria of SRP 15.3.3, “Reactor Coolant Pump Rotor Seizure,” related to the peak pressure and is, therefore, acceptable.

3.1.2 Loss of External Load or Turbine Trip

For a loss of external load or a turbine trip event, the licensee reanalyzed four cases with both minimum or maximum reactivity coefficients under conditions where pressurizer spray (PS) and power operated relief valves (PORVs) were assumed to be actuated or inoperable. The two cases that assumed actuation of PS and PORVs modeled the pressurizer safety valves (PSVs) as having a -3 percent setpoint tolerance, whereas the two cases that assumed PS and PORVs that were inoperable modeled a +3 percent setpoint tolerance for the PSVs. The licensee performed the reanalysis with the current licensing-basis LOFTRAN code. The plant characteristics and initial conditions assumed in the reanalysis were the same as those currently assumed in the licensing-basis analysis. Nominal values were assumed for the initial reactor power, temperature, and pressure. The assumptions of the nominal values for the plant’s initial conditions are consistent with the NRC-approved Westinghouse revised thermal design procedure used in the current licensing-basis analysis.

The licensee used current licensing-basis methods to analyze this event and showed that the calculated peak RCS pressure is below 110 percent of the RCS design pressure and the lowest DNBR does not exceed the safety DNBR limit. The staff finds that the results satisfy the acceptance criteria of SRP 15.2.1 and 15.2.3, “Loss of External Load and Turbine Trip.” Therefore, the staff concludes that the reanalysis is acceptable.

3.1.3 Loss of Normal Feedwater Flow (LONF) and Loss of All AC Power to the Plant Auxiliaries (LOAC)

The licensee indicated that the analysis of record for the LONF and LOAC events assumed the PORVs to be operable during the transients. As a result, the pressurizer pressure did not increase to a value that could be influenced by the proposed PSV setpoint tolerance changes. Therefore, the increase in the as-found PSV setpoint to $\pm$3 percent does not affect the plant response predicted in the analysis of record.

In addition, the transient experienced during an LONF or an LOAC is bounded by the current loss of load licensing-basis analysis, since the turbine trip for the LONF and LOAC transients occurs after reactor trip. The delayed turbine trip provides additional heat removal via steam flow to the turbine, limiting an increase in the RCS pressure. Following the turbine trip, the auxiliary feedwater system is capable of returning the plant to a safe condition by removing the stored and residual energy heat, thus, preventing either overpressurization of the RCS or uncovery of the core. As discussed in Section 3.1.2, the reanalysis of the bounding event and the loss of load event, with an assumption of ±3 percent setpoint tolerance for the PSVs, showed that the peak pressure is within the design pressure safety limits and the calculated minimum DNBR is above the safety limit value. Since the LONF and LOAC events are bounded by the loss of load event, they will continue to meet the design pressure and DNBR safety limits. Therefore, the staff concludes that the results of the LONF and LOAC events with an assumed increase in the as-found PSV setpoint tolerance to ±3 percent are acceptable.

3.1.4 Feedwater Line Break

A sensitivity study was performed for the feedwater line break event using the licensing basis LOFTRAN code. The system parameters and initial conditions assumed in the reanalysis were the same as those currently assumed in the licensing-basis analysis. The sensitivity study assessed the effects of increasing the PSV setpoint tolerance to ±3 percent. The reanalysis confirmed that adequacy of the results presented in the UFSAR. Therefore, the staff concludes that the increase in the as-found PSV setpoint tolerance to ±3 percent will not affect the analysis of main feedwater line break event, and the analysis of record remains acceptable.

3.1.5 Safety-Related Valve Performance

For overpressure events, the licensee verified that the effects of the increased setpoint tolerance on the performance of safety-related valves have been evaluated. The licensee determined that the increased TS PSV setpoints were considered in ensuring the ability of the applicable valves to perform their safety function. The staff has determined that this is acceptable because the valves are capable of performing their safety functions at the higher setpoint pressure.

3.2 TSs 3.4.2 and 3.4.3 - Safety Valves - Shutdown and Operating for Unit 2

Current TS 3.4.2, "Safety Valves - Shutdown," and TS 3.4.3,"Safety Valves - Operating," for Unit 2 require that all PSVs be operable with a lift setting of 2485 psig ±1 percent. The licensee proposed to change the PSV setpoint tolerance from the current requirements of ±1 percent of 2485 psig to ±3 percent for Unit 2. Since the TS changes adequately reflect the results of the acceptable safety analysis (discussed in Section of 3.1) and they are consistent with Westinghouse STS, Base SR 3.4.10.1, which allows the as-found setpoint tolerance for PSVs to increase up to ±3 percent, the staff concludes that the TS changes related to the as-found PSV setpoint tolerance are acceptable.

The licensee also proposed to add an as-left PSV setpoint tolerance of ±1percent of 2485 psig to TS 3.4.2 and TS 3.4.3 for Unit 2. The staff determined that the TS changes are acceptable since the changes are consistent with Westinghouse STS, Base SR 3.4.10.1, which allows up to ±1 percent as-left setpoint tolerance for the PSVs.

3.3 TSs 3.4.2 and 3.4.3 - Safety Valves - Shutdown and Operating for Unit 1

Current TS 3.4.2, "Safety Valves - Shutdown," and TS 3.4.3,"Safety Valves - Operating," for Unit 1 require that all PSVs be operable with a lift setting of 2485 psig ±3 percent. The licensee proposed to add an as-left PSV setpoint tolerance of ±1percent of 2485 psig for Unit 1. The proposed ±1percent as-left setpoint criterion is consistent with Westinghouse STS, Base SR 3.4.10.1. Therefore, the staff concludes that the TS changes are acceptable.

4.0 SUMMARY

Based on the considerations discussed in Section 3.0 above, the staff has concluded that the proposed TSs relating to the as-found and as-left setpoint tolerances for the PSVs on D. C. Cook, Unit 2, and the as-left setpoint tolerance for the PSVs on D. C. Cook, Unit 1 are acceptable, since (1) the proposed setpoint tolerances adequately reflect the results of the acceptable transient analysis and are consistent with the Westinghouse STS, Base SR 3.4.10.1; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (3) such activities will be conducted in compliance with the Commission's regulations and issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (67 FR 15624). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: G. Hammer
S. Suns

Date: November 26, 2002